10 CFR Part 63 – Information to Support Safety Decisions

Tim Kobetz
Mahendra Shah
Robert K. Johnson

July 20, 2005

Enclosure 3

Overview

- Past Interactions on the LA Content
- 10 CFR Part 63 Regulatory Framework
- LA Content
- LA Content - Examples
- Summary

Past Interactions on the LA Content

- February 2004 Technical Exchange
- May 2004 Technical Exchange
- September 2004 Technical Exchange
- October 8, 2004 Letter from NRC to DOE
- June 2005 Technical Exchange

10 CFR Part 63 Regulatory Framework

- Licensing Approach of 10 CFR Part 63
  - One license application (LA) for Part 63 with two regulatory decisions
    - 10 CFR 63.31 - Whether to grant a Construction Authorization based on the LA
    - 10 CFR 63.41 - Whether to grant a license to receive and possess source, special nuclear, or byproduct material, based, in part, on the substantial construction completion in conformity with the LA as amended, and updated LA for additional data obtained during construction

10 CFR Part 63 Regulatory Framework (contd.)

Construction Authorization - 10 CFR 63.31

- In reaching findings related to safety, the staff will consider, in part:
  - The DOE has described the proposed geologic repository as specified at § 63.21
  - The site and design comply with the performance objectives and requirements contained in Subpart E *(Technical Criteria)* of this part

LA Content

- 10 CFR 63.21
  - (c) The Safety Analysis Report must include:
    - (3) A description and discussion of the design of the various components of the geologic repository operations area and engineered barrier system including:
      - (i) Dimensions, material properties, specifications, analytical and design methods used along with applicable codes and standards;
      - (ii) Design criteria used and their relationships to the preclosure and postclosure performance objectives specified at § 63.111(b), § 63.113(b), and § 63.113(c); and
      - (iii) The design bases and their relation to the design criteria.
    - (5) A preclosure safety analysis…to ensure compliance with § 63.111(a), as required by § 63.111(c).

LA Content (contd.)
Performance Objectives (§ 63.111)

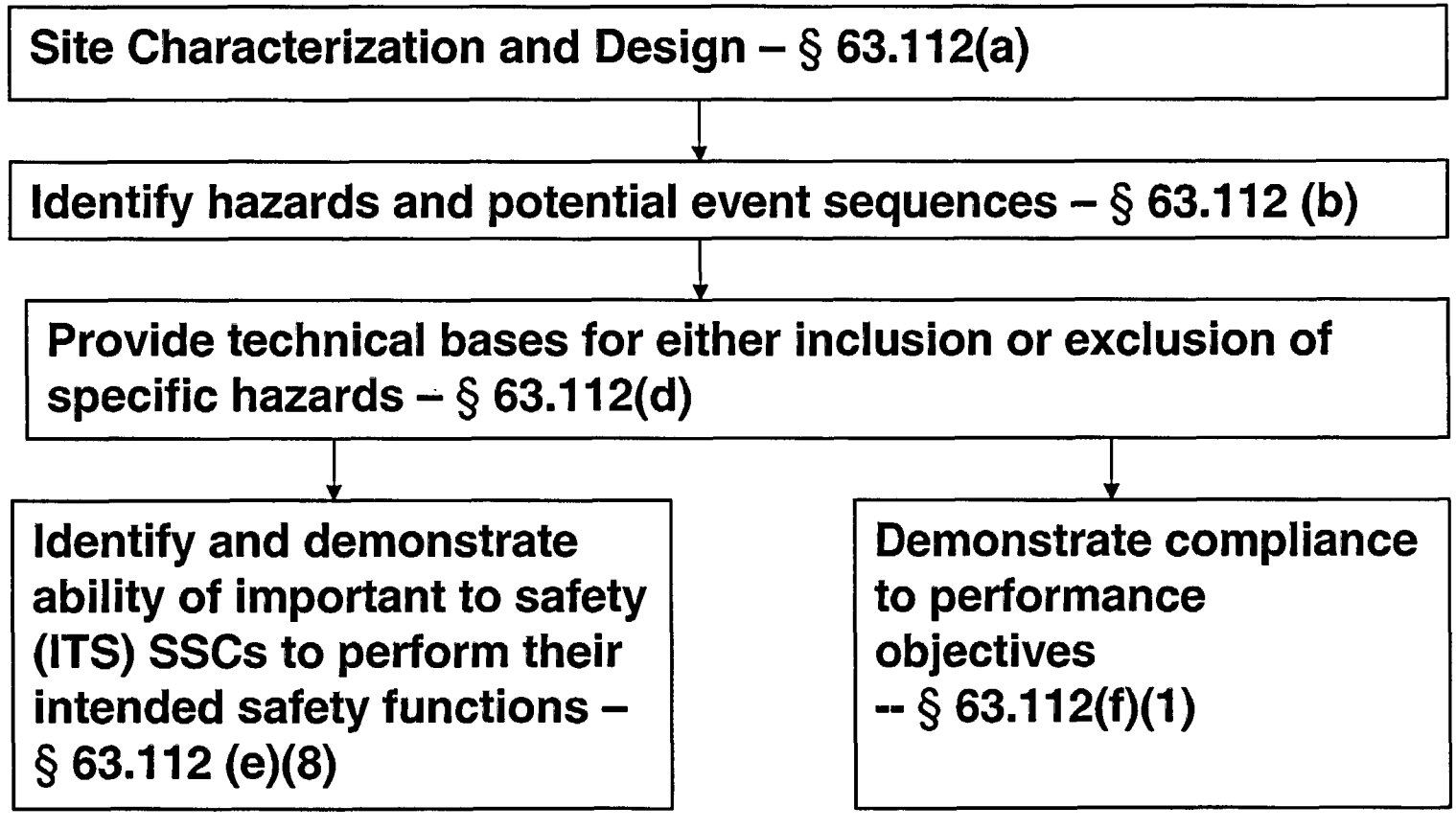
- PCSA must be performed to demonstrate dose performance objectives are not exceeded.
  - 10 CFR Part 20 for Normal and Category 1 events (workers)
  - 15 mrem/yr for Normal and Category 1 events (public)
  - 5 rem/event for Category 2 events (public)

LA Content (contd.)

Preclosure Safety Analysis (PCSA)

- 10 CFR 63.112 (in Subpart E)
  - (e) An analysis of the performance of structures, systems, and components to identify those that are important to safety…The analysis required in this paragraph must include…
    - (8) Ability of structures, systems, and components to perform their intended safety functions, assuming the occurrence of event sequences.
  - (f)(1) The relationship between design criteria and the requirements specified at § 63.111(a) and (b); and
  - (f)(2) The design bases and their relation to the design criteria.

LA Content (contd.)
Preclosure Safety Analysis (PCSA)



LA Content – Examples Overview

- Information to Support the Safety Decisions
- Aircraft Crash Hazard
- Trolley
- Spent Fuel Transfer Machine
- Concluding Remarks

LA Content – Examples
Information to Support the Safety Decisions

For SSCs that are important to safety, the information to support the safety decisions under § 63.31 and § 63.41 is the demonstration of their ability to perform intended safety functions [63.112(e)(8)]. The DOE may choose to demonstrate this with the following:

- **Structural integrity evaluation**
- **Systems performance reliability**

LA Content – Examples
Aircraft Crash Hazard

Aircraft Crash has been identified as a potential human-induced hazard at the proposed geological repository operations area, requiring evaluation for compliance with 10 CFR 63.112 requirements.

LA Content – Examples
Aircraft Crash Hazard (contd.)

DOE Approach based on the Technical Exchange in June, 2005

Identify Aircraft Crash event frequencies, aircraft type and velocities

↓

Reduce the crash event frequencies for No-fly Zone, and the ability of the exterior building walls and the barriers surrounding the aging facilities to withstand the aircraft crash.

↓

Remaining Crash event frequencies are less than 1 in 10,000 for the preclosure period (10 CFR 63.2, Event Sequence, Category 2). Therefore, these events are screened out and not included in § 63.111(b)(2) evaluation.

LA Content – Examples
Aircraft Crash Hazard (contd.)

Information needed to verify the ability of the walls and barriers to perform their intended safety functions for the aircraft crash events:

- Structural Evaluation of walls and supported systems/components, and the barriers, for
  - Direct impact loads
  - Thermal effects due to potential fire

LA Content – Examples
Aircraft Crash Hazard (contd.)

Information required for the direct impact loads Structural Integrity Evaluation:

- Physical details of building walls/foundations (essential elements), Barriers (dimensions, material properties, etc.)
- Aircraft Type and Impact Speed, including the bases for their selection
- Method of calculating the impact force time-histories
- Impact locations analyzed, and their bases

LA Content – Examples
Aircraft Crash Hazard (contd.)

Information needed for the direct impact loads (contd.)

- Local (penetration) and Global (overall structural response) analyses methods and results, stability of building/foundation, including computer codes, if used.

- The global evaluation must consider all event sequences associated with the aircraft crash until the event is finished, including the effects on the systems within the building, and the potential for release of radioactivity for evaluating compliance with 10 CFR 63.111(a) and 63.111(b).

LA Content – Examples
Aircraft Crash Hazard (contd.)

Information required for evaluation of thermal effects due to potential fire:

- Fire Parameters, including the maximum duration and temperature profiles (temporal and spatial), computer codes, if used.

- Structural Evaluation of the walls, affected SSCs, and the Barriers, for thermal effects

LA Content – Examples Trolley

- Potential Hazards
  - Derailment, tipover, or collision of trolley
- Potential Event Sequences
  - Drop or damage to canister or WP
  - Failure of canister or WP containment
  - Release of SNF particulates into transfer area
- ITS and Related SSCs
  - Trolley structure and components credited with preventing a drop
  - DOE indicates a drop is beyond Category 2 because it is prevented by design

LA Content – Examples
Trolley (contd.)

- Information to Support the Safety Decisions
  - Design and operational aspects, as appropriate (e.g., critical dimensions, movements)
  - Technical basis for ability of trolley not to tipover, derail, or drop its load in the event of a collision at the maximum speed (e.g. reliability and controls assessment)

LA Content – Examples
Spent Fuel Transfer Machine

- Potential Hazards
  - Potential drop or collision involving SNF assemblies
- Potential Event Sequences
  - Drop or collision of SNF
  - Drop or collision of handling equipment onto SNF
  - Release of SNF particulates into transfer room
- ITS and Related SSCs
  - Spent fuel transfer machine components (e.g., grapple)
  - Primary confinement HVAC system
  - DOE has identified failures as Category 1 (collisions/drops) and Category 2 (handling drop), based on design and operational requirements

LA Content – Examples
Spent Fuel Transfer Machine (contd.)

- Information to Support the Safety Decisions
  - Design and Operational aspects, as appropriate (e.g. lift and grapple mechanisms, motion control)
  - Technical basis for design and operational reliabilities such as 10^{-5} drop and collision rates, and 10^{-7} handling equipment drop rate (including human reliability).

LA Content – Examples Conclusions

- The examples we discussed illustrate the types of information that should be in LA to support the safety decisions:
- The information in LA (design and supporting analyses that provide the technical bases) should be to sufficient depth to demonstrate the ability of ITS SSC to perform their intended safety functions, and meet the 63.111(a) and 63.111(b) requirements.
- Design and operational requirements alone may not be enough to "demonstrate ability of SSCs to perform their intended safety functions". Design specification and supporting analyses should be to sufficient depth to demonstrate performance.

Summary

- 10 CFR Part 63 is a one license application (LA) with two regulatory decisions:
  - A decision on Construction Authorization (§ 63.31) is based, in part, on the safety of the proposed design. The LA must contain information to sufficient depth for ITS SSCs to demonstrate their ability to perform safety functions.
  - A decision on the License to Receive and Possess (§ 63.41) is based, in part, on the substantial construction completion in conformity with the LA as amended, and updated LA for additional data obtained during construction.

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Opening Remarks

Presented to:
NRC/DOE Technical Exchange on Information to Support 10 CFR Part 63 Analyses

Presented by:
Joseph D. Ziegler
Director, Office of License Application and Strategy
Office of Repository Development
U.S. Department of Energy

July 20, 2005
Las Vegas, Nevada

License Application Requirements and Guidance

- Content in accordance with 10 CFR 63.21
- Addresses the guidance in NUREG-1804, Yucca Mountain Review Plan
- Safety analyses and supporting information that form the basis for the License Application (LA) and meet applicable Quality Assurance (QA) requirements
- Use of NRC guidance, precedent and consensus industry codes and standards, supported by analysis, as needed
- DOE management controls and processes ensure high quality of the LA

LA Development Process

- **Content**
  - Contains General Information and Safety Analysis Report
  - Extracted from underlying design and analysis documents
  - Presents technical justification for preclosure and postclosure nuclear safety basis
  - Provides sufficient detail to identify "Important to Safety" structures, systems, and components (SSCs)
  - Demonstrates that the design and operations will meet performance objectives
- **Supporting Documents**
  - Prepared in accordance with applicable QA requirements
  - Examples include scientific reports, calculations, analyses, drawings, facility and system description documents, project functional and operational requirements, and project design criteria
  - Detail information needed to support construction authorization decision per 10 CFR 63.31 will be in the License Application

LA Development Process

- **Series of iterative reviews to ensure:**
  - **Completeness and accuracy**
  - **Consistency throughout LA**
  - **Consistency with supporting documents**
  - **Adequate descriptions of the safety basis**
- **Series of senior internal management and senior nuclear experts reviews to:**
  - **Independently assess the safety basis**
  - **Determine readiness of LA for submittal**

DOE Regulatory Bases to Support NRC Decision to Issue a Construction Authorization

- The LA is complete as possible in light of information that is reasonably available at the time of docketing pursuant to 10 CFR 63.21(a)
- The LA includes sufficient information to demonstrate that the:
  - Design will ensure that the waste can be safely handled and emplaced
  - Processes and controls will ensure that safety requirements are met
- The LA documents the analytical bases that demonstrate compliance with the pre-closure technical criteria in 10 CFR 63 Subpart E. This includes and/or is supported by:
  - Design information that covers the facilities and inventory in sufficient detail to support the pre-closure analyses
  - Use of consensus industry standards translated into design requirements for structures, systems, and components important to safety and important to waste isolation
  - Design inputs and physical parameters with appropriate justification, supporting an analytical safety basis
  - Description of management controls
  - Procedural safety controls which are credited in the safety analysis
  - Combination of detailed calculations and bounding/representative analyses

Change Control for Documents Supporting the Licensing Basis

- Potential changes as a result of:
  - Request for Additional Information (RAI) process
  - Design changes to incorporate:
    - New technologies
    - New information obtained during construction
  - Safety Evaluation Report (SER) Open or Confirmatory Items
  - Contentions
- Processes and controls to:
  - Control design and safety basis changes
  - Assess potential impacts to the safety basis from changes
  - Ensure compliance with applicable QA requirements
- Future changes to design will be evaluated to determine if a change to the safety analysis is required

DOE's Proposed Dates for DOE/NRC Preclosure Technical Exchanges

- Waste Package Transporter and Gantry / Material Handling / Non-standard Equipment – August 2005
- Aging / Structural Analyses for Aircraft Hazards – August 2005
- Pre-closure Consequences and Worker Doses – September 2005
- Fuel Behavior and Release Fractions – September 2005
- Pre-closure Criticality – September 2005
- Pre-closure Seismic Design – October 2005
- Design and Classification of Electrical Systems – October 2005
- CSNF Handling in a Dry Environment – October 2005
- Technical Specifications – November 2005

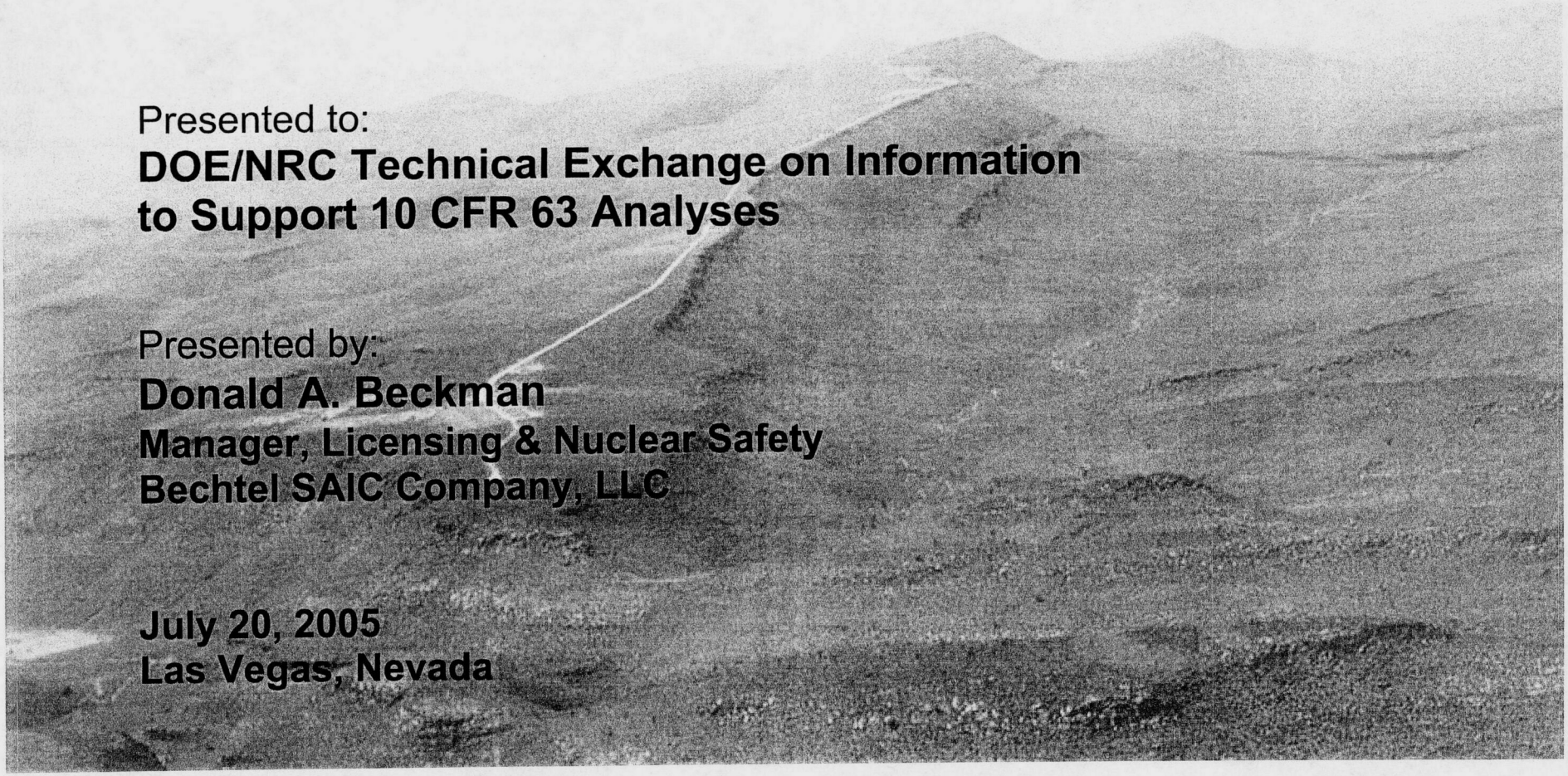
Summary

- The LA will meet statutory and regulatory requirements
- Supporting detail will be available in technical documents
- Supporting documents will be available for NRC inspection

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Availability of Information



Presented to:
DOE/NRC Technical Exchange on Information to Support 10 CFR 63 Analyses

Presented by:
Donald A. Beckman
Manager, Licensing & Nuclear Safety
Bechtel SAIC Company, LLC

July 20, 2005
Las Vegas, Nevada

Availability of Information

- **Purpose**
  - Discuss the information to be contained in the License Application (LA) and the bases for demonstrating that safety requirements have been appropriately identified and implemented into the design to ensure compliance with 10 CFR 63 requirements and support NRC's safety determination in support of Construction Authorization (CA)

Availability of Information

- **Nuclear Safety Design Basis (NSDB) Hazards Examples:**
  - **Seismic**
    - **Equipment (Standard vs. non-Standard)**
    - **Structural**
  - **Load Drop**
    - **Waste Package**

Availability of Information

Standard vs. non-Standard Equipment

- **Standard:**
  - The majority of components have been assessed as being sufficiently similar to typical industry designs, based on consensus codes and standards (C&Ss), that no "special" design development needs have been identified
- **Non-Standard:**
  - For the small minority of components that were initially assessed as not being sufficiently similar to typical industry designs, additional process steps are followed:
    - Confirm the initial assessment (gap analysis), and if confirmed,
    - Identify the "design needs" that will ensure a fabricated component can meet its functional and safety (NSDB) requirements
    - To date, only 10 components have been identified with supplemental design needs and these have been documented in design development plans (DDPs)

Availability of Information

Seismic – "Standard" Equipment Example

- NSDB requirement:
  - No seismically induced failure
- Application:
  - Piping, duct supports
- Analysis approach:
  - Standard engineering practices
    - Standard Equipment addressed by C&Ss
    - Consistent with previously licensed nuclear industry application/practices
- Basis for support of LA
  - Combination of consensus C&Ss, historical data, design precedent, and proven technology and methods ensure adequate analysis and design

Availability of Information

Seismic – "non-Standard" Equipment Example

- NSDB Requirement – Mechanical Handling Equipment:
  - No tip-over or contact
  - No container/canister breach
- Application:
  - Cranes, Trolleys, SNF Handling Machine, Waste Package Tilting Machine
- Analysis Approach:
  - Gap analysis to determine extent of consensus C&S coverage
  - Bounding, scoping or similarity analyses ensure viability of design to meet safety requirements
  - Design development plan identifies in detail additional design/fabrication/testing/analysis requirements for vendor
- Basis for Support of LA
  - Same as for "Standard" with the addition of bounding, scoping, or similarity analyses, gap analysis, and DDPs, ensure adequate analysis and design

Availability of Information

Seismic – Structural Example

- **NSDB Requirement:**
  - No failure causing radiological release
- **Application:**
  - Concrete and steel structures
- **Analysis Approach:**
  - Bounding Analysis
    - Consensus Codes & Standards combined with a bounding analysis to verify viability of design
- **Basis for Support of LA**
  - Project specific evaluation incorporating detailed site seismic data using proven technology and methods to ensure adequate analysis and design

Availability of Information

Load Drop – Waste Package Example

- **NSDB Requirement:**
  - No breach
- **Application:**
  - Waste Packages
- **Analysis Approach:**
  - Detailed structural analysis to identify design limits
- **Basis for support of LA**
  - Project specific detailed analyses using proven technology and methods ensure adequate analysis and design

Availability of Information

NSDB Hazard	NSDB Requirement	NSDB Req. Applied to:	Analysis Approach	Adequacy to Support LA
Seismic - Structural	· No Failure with Radiologic Release	· Structural Concrete and Steel	**Analyses** C&Ss combined with a bounding analysis to verify Project applicability. · Code Analysis using Standard Review Plan (NUREG-0800). · Concrete Code ACI-349. · Structural Steel Code AISC-N690.	· Project specific evaluation incorporating detailed site seismic data.
Seismic - Equipment	· No seismically induced failure	· Piping, Duct Supports	**Standard Engineering Practices** · Standard equipment addressed by C&Ss. · ASME B31.3.	· C&Ss provide assurance through history of adequate design.
	· No tip-over or contact · No Breach	· Cranes, Trolleys, Spent Fuel Handling Machine, Waste Package Tilting	**Standard Engineering Practices** · ASME NOG-1-2002 for lifting equipment and equipment with similar features. · Combination of AISC Manual of Steel Construction (1997) and ASCE-4-98 for other equipment. **Analyses** Bounding Analysis, Gap Analysis, and Design Development Plan (DDP) for Non-Standard equipment (C&Ss not entirely applicable).	· C&Ss provide assurance through history of adequate design. · Bounding Analysis. · Gap Analysis and DDPs provide for adequate analysis and design.

Availability of Information

NSDB Requirement	NSDB Req. Applied to:	Analysis Approach	Adequacy to Support LA
· No Breach	· Waste Packages	**Analyses** · Detailed structural analysis to identify design limits.	· Detailed analyses per accepted methods.
· 1 x 10E-05 per Lift	· Cranes	**Standard Engineering Practices** · Historical failure rate data and C&Ss. · ASME NOG-1-2002.	· Similar design & applicability of industry C&Ss and incorporates historical data.
· No Slapdown	· Tilting Machine	**Analyses** · Bounding Analyses, Gap Analysis and DDP for Non-Standard equipment.	· Design and construct to prevent slapdown.
· Confinement Boundary	· Passive facility design	**Standard Engineering Practices** · Maintain negative pressure. Passive Design features per C&Ss. · ASME AG1 Code on Nuclear Air and Gas Treatment.	· Assure negative pressure (no out leakage).
· Mitigate Release	· Heating, Ventilation and Air Conditioning (HVAC)	**Standard Engineering Practices** · Two-stage HEPA filtration (99% DF each). · ASME AG1 Code on Nuclear Air and Gas Treatment. **Analyses** · Bounding Analysis · Minimum operations reliability (4 Hrs. w/o failure).	· DOE Air Cleaning Handbook incorporating industry standards. · HEPA Filter testing and industry use. · PCSA conclusion that 4 Hrs. provides reasonable assurance for mixing and removal.

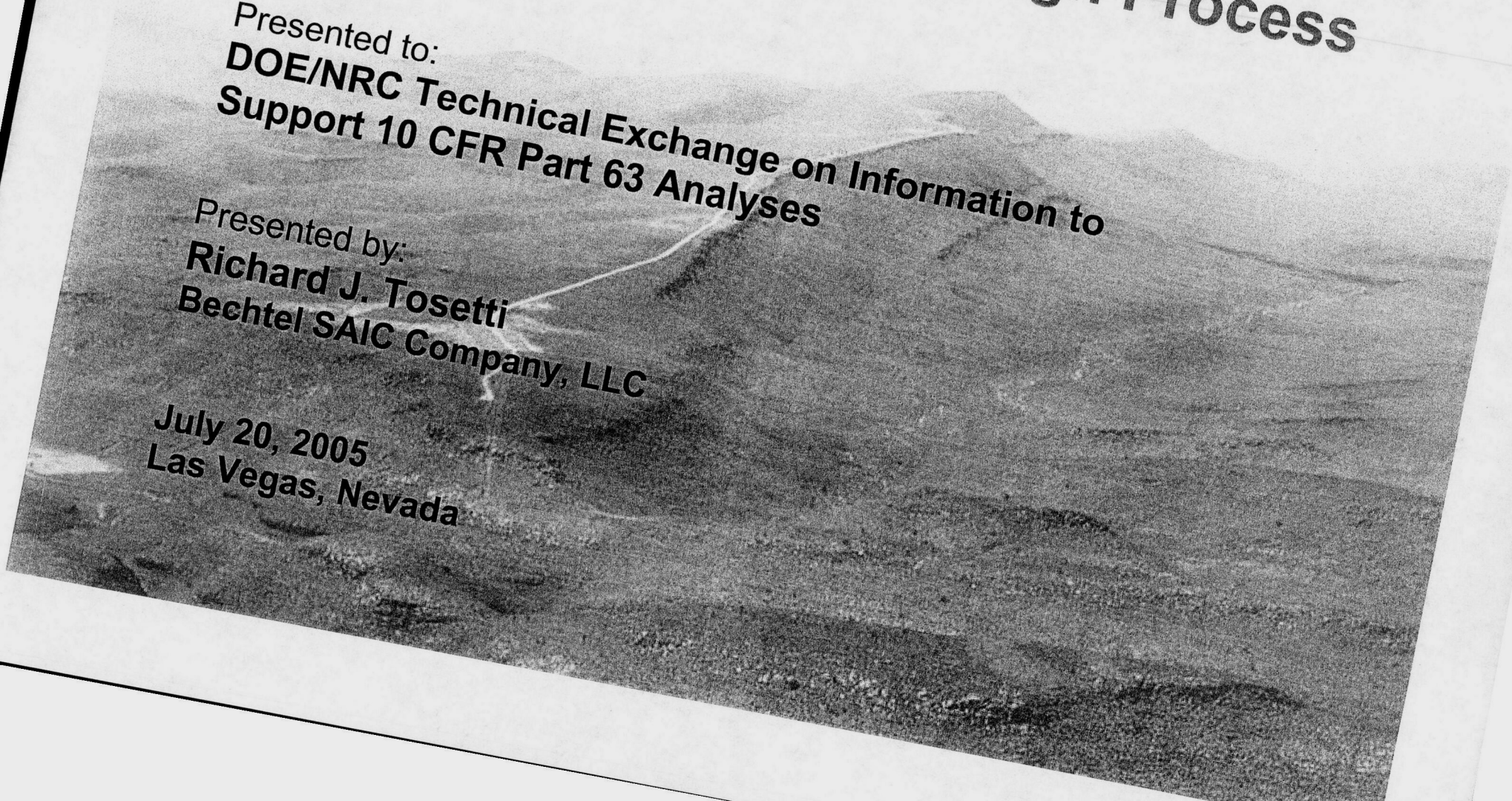
Availability of Information

- **The LA and supporting documentation at time of submittal will be sufficient to demonstrate that safety requirements have been appropriately identified and implemented into the design to ensure compliance with 10 CFR 63 requirements and support NRC's safety determination in support of CA**
  - **Full set of surface facilities**
    - **Expected operations and hazards**
    - **Maximum throughput**

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Functional Discipline Design Process

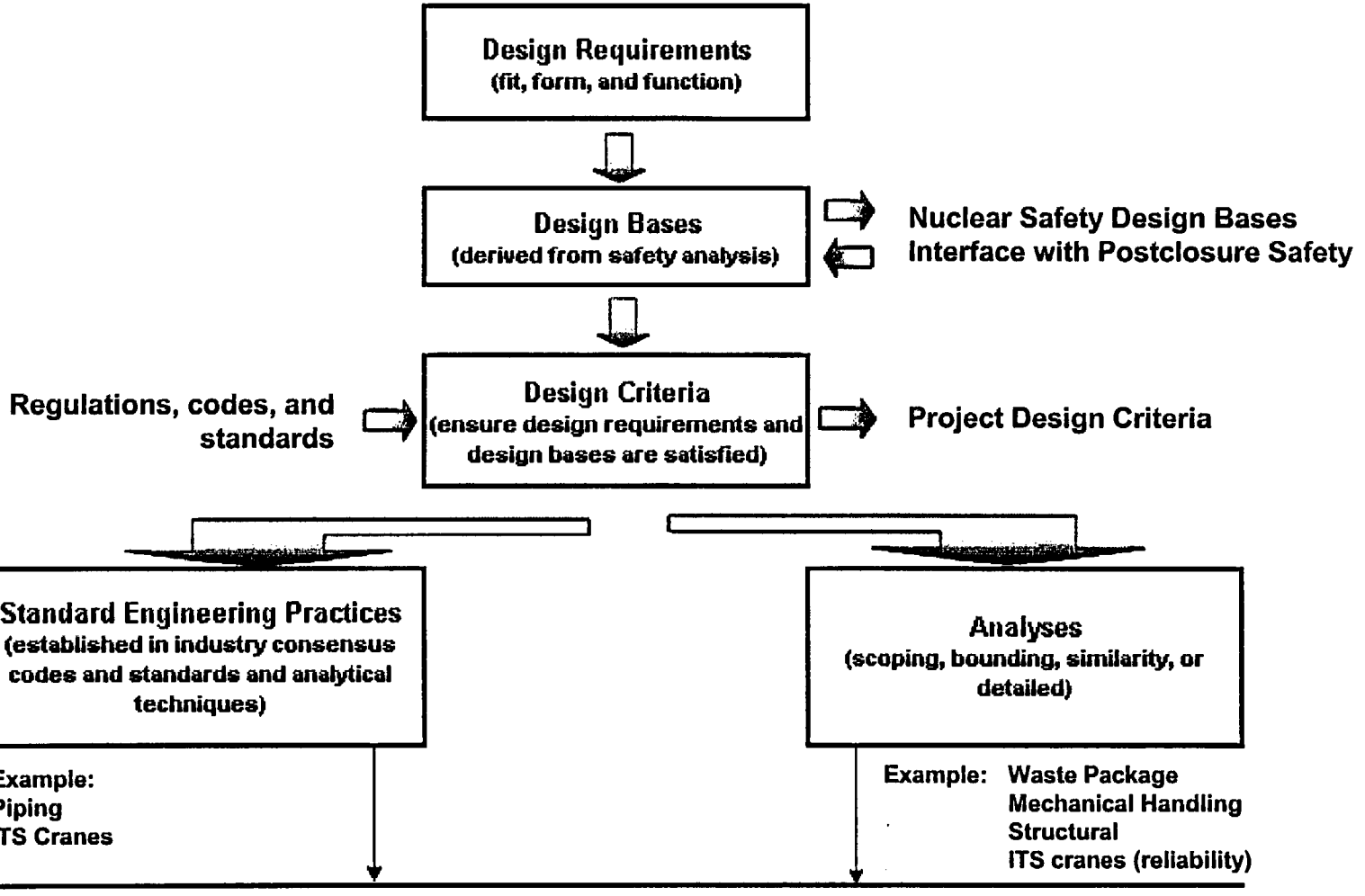


Presented to:
DOE/NRC Technical Exchange on Information to Support 10 CFR Part 63 Analyses

Presented by:
Richard J. Tosetti
Bechtel SAIC Company, LLC

July 20, 2005
Las Vegas, Nevada

Information Available Process Map



*ITS = Important to Safety

Design and/or Analysis Defined by Code/Standard

- These activities are where a consensus code or standard is called out that defines the process
  - Piping — The American Society of Mechanical Engineers (ASME) B31.3 - Process Piping
  - HVAC — American National Standards Institute (ANSI) / American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) ASME - AG1 - Code on Nuclear Air and Gas Treatment
  - Concrete — American Concrete Institute (ACI) 349 - Code Requirements for Nuclear Safety Related Concrete Structures

    ACI 318 - Building Code Requirements for Structural Concrete and Commentary

Design and/or Analysis Defined by Code/Standard
(continued)

- **These activities are where a consensus code or standard is called out that defines the process** (continued)
  - **Electrical** — **Institute of Electrical and Electronic Engineers (IEEE) Std. 141-1993 - Recommended Practice for Electric Power Distribution for Industrial Plant IEEE Std. 446-1995 – Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications**

Piping Design

Design Requirements

Sources

Basis 10 CFR 63 (Risk Informed Performance Based Design)
Interface Requirements (Facility and System Description Documents)
Functional & Operational Requirements (F&OR)

Design Bases

Documents

Nuclear Safety Design Basis

Design Criteria

Documents

ASME B31.3
Project Design Criteria

Standard Engineering Practice

Description

ASME B31.3 clearly describes the process for performing piping stress analysis. This code covers all aspects of piping design.

For example ASME B31.3 describes:

Conditions and Criteria
- Design pressure and temperature
- Ambient effects, dynamic effect, thermal effects etc.
- Allowable stresses and other stress limits

Pressure design of Piping Components
- Pressure design of components, such as pipe, fittings, flanges, etc.

Fluid Service Requirements for Piping Components and Piping Joints
- General and specific requirements of welded joints and mechanical joints

Flexibility an Support
- Properties for Flexibility Analysis
- Calculation of Movements

Systems
- Specific piping systems
- Pressure relieving systems

- By committing to ASME B31.3 the analysis is bounded
- Analysis is not required for submittal of LA

Detailed Analysis

- **Detailed analysis describes the actual structure, system, and component (SSC)**
  - **For example: The waste package has an existing suite of analyses that describes the functions of the waste package (structural, thermal, shielding and criticality). These functions and the applicable codes have been set forth within a technical specification that defines the requirements for fabrication and assembly of the prototype waste package for the vendor**
  - **The 21-Pressurized Water Reactor (PWR) suite of documents has been selected as an example as it represents over 50 percent of the commercial spent nuclear fuel (CSNF) waste packages and has the highest heat output and source term**

Example of 21–PWR Analysis Documents

- **Structural**
  - 21-PWR Waste Package with Absorber Plates Fabrication Specification
  - Object Drop on 44-Boiling Water Reactor (BWR) and 21-PWR Waste Packages
  - 21-PWR Tip Over from Elevated Surface
  - 21-PWR Waste Package 10-Degree Oblique Drop with Slap Down
  - 21-PWR Waste Package Drop with Emplacement Pallet
  - Design and Engineering, 21-PWR Waste Package Configuration
  - Horizontal Drop of the 21-PWR Waste Package Supplemental Calculation
  - Structural Calculations of Waste Packages Exposed to Vibratory Ground Motion
  - Vertical Drop of 21-PWR Waste Package with Lifting Collars
  - Corner Drop of the 21-PWR Waste Package
  - Swing Down Drop of the 21-PWR Waste Package

Example of 21–PWR Analysis Documents

(Continued)

- **Thermal**
  - Waste Package Outer Barrier Stress due to Thermal Expansion with Various Barrier Gap Sizes
  - Thermal Evaluation of a 21-PWR Waste Package in the Shielded Surface Facility Transporter
  - Thermal Response of the 21-PWR to a Fire Accident
- **Shielding**
  - Dose Rate Calculation for 21-PWR Waste Package
- **Criticality**
  - 21-PWR Waste Package with Absorber Plate Loading Curve Evaluation
- **Technical specification**
  - 21-PWR Uncanistered Fuel (UCF) Prototype Fabrication Specification

Bounding Analysis

- **Two types of example of how the design is bounded**
  - **Mechanical Handling**
    - **The design is bounded by performing analyses in conjunction with codes and standards**
  - **Structural**
    - **The design is bounded by using conservative estimates or assumptions**

Mech. Handling Design of ITS Equipment

Design Requirements

Sources

Basis 10 CFR 63 (Risk Informed Performance Based Design)
Regulatory Guidance (NUREG-0554, *Single-Failure-Proof Cranes for Nuclear Power Plants*, NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, NUREG-1804, Revision 2, *Yucca Mountain Review Plan*)
Interface Requirements (Facility and System Description Documents, Waste Acceptance System Requirements Document)
Functional & Operational Requirements

Design Bases

Documents

Nuclear Safety Design Basis

Design Criteria

Documents

Project Design Criteria

Standard Engineering Practice

Documents

For standard equipment, relevant codes and standards are applied, for non-standard, codes and standards are applied at the component level.
Systems described in the System Description Documents (SDDs)
- SDDs identify all requirements (flowdown from upper tier sources of requirements)
- Describe system (conceptual at LA)
- Describe system operation

System functions delineated on Block Flow Diagrams (BFDs, Level 1, 2 & 3)
- Detailed functions of mechanical handling systems
- Sequential, in order performed
- Type of equipment added at Level 3

Process & Instrument Diagrams (P&IDs)
- Identify instrumentation and controls, including vendor and system interfaces
- Identify high-level logic requirements

Equipment Specifications
- Identify equipment-specific requirements from applicable SDDs, including performance, safety, configuration, interface, analytical, and testing
- Invoke relevant codes and standards*, Quality Assurance (QA) requirements

Bounding Analysis

Documents

Typical Bounding Analyses
- Such as sizing, reliability, and seismic loading analyses

For each category of important-to-safety equipment identified in the system documents, a preliminary design is developed and described on the following documents:

Mechanical Equipment Envelopes (MEEs) or Data Sheets
- Identify size, spatial requirements, and limiting parameters (e.g. minimum capacity)
- Identify facility and equipment interface requirements

*For "non-standard" equipment, supplemental and design development activities are identified for areas of the design not adequately addressed by industry codes or standards. These activities are performed by the project or the equipment vendor.

Structural Design of Nuclear Facilities

Design Requirements

Sources

Basis 10 CFR 63 (Risk Informed Performance Based Design)
Regulatory Guidance (NUREG-0800, *Standard Review Plan for the Review of Safety Analysis for Nuclear Power Plants*, Codes and Standards for ITS Structures for Loads)
Interface Requirements (Facility and System Description Documents)
Functional & Operational Requirements

Design Bases

Documents

Nuclear Safety Design Basis
Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain
Seismic Analysis & Design Approach Document
Seismic Input Report

Design Criteria

Documents

Load Combinations and Acceptance Criteria
Concrete Design based on ACI-349 Code
Structural Steel Design based on the American Institute of Steel Construction (AISC) - N690 Code
Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities
Project Design Criteria

Bounding Analysis

Analysis Description

- Design performed using loads such as deal load, live loads and loads from Internal and External Hazards
- Load combinations considered are based on NUREG-0800 requirements
- The seismic loads are based on site specific Design Basis Ground Motions (DBGMs) defined in terms of Ground Acceleration Response Spectra for the design of ITS surface facilities
- The surface facilities are designed for DBGM2 (Mean Annual Probability Exceedance of 5×10^{-4}) and evaluated for Beyond Design Basis Ground Motion (BDBGM) (Mean Annual Probability Exceedance of 1×10^{-4})
- Bounding analysis for seismic loads uses a lumped mass multiple stick model idealization to represent the structural configuration. This approach was developed based on licensing precedence from Nuclear Power Plants
  - The foundation impedance is considered by use of "Soil Spring" approach as per ASCE 4-98
  - Comparison of the base shear obtained from the lumped mass "Soil Spring" vs. System for Analysis of Soil-Structures Interaction (SASSI) shows the spring provides more conservative estimates
- Design will include an aircraft impact analysis which is currently underway
- Bounding analyses uses conservative estimates and methods to establish design feasibility to site specific seismic conditions
- Detailed Design will be based on Industry Codes and Standards

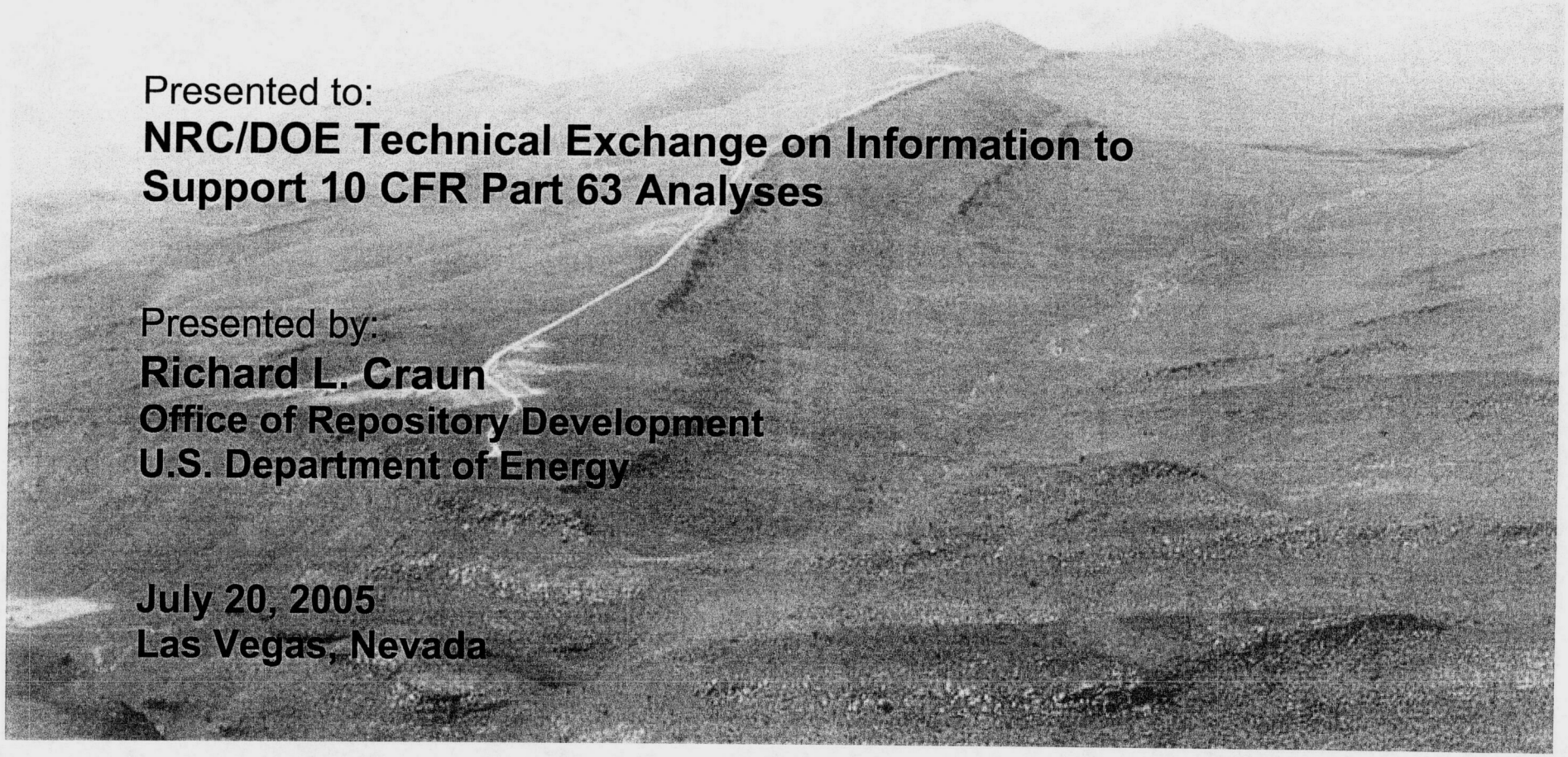
Summary

- By following the process of using bounding analyses and the consensus codes and standards, it is shown that the Nuclear Safety Design Bases are met

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Functional Discipline Design Process – Overview



Presented to:
NRC/DOE Technical Exchange on Information to Support 10 CFR Part 63 Analyses

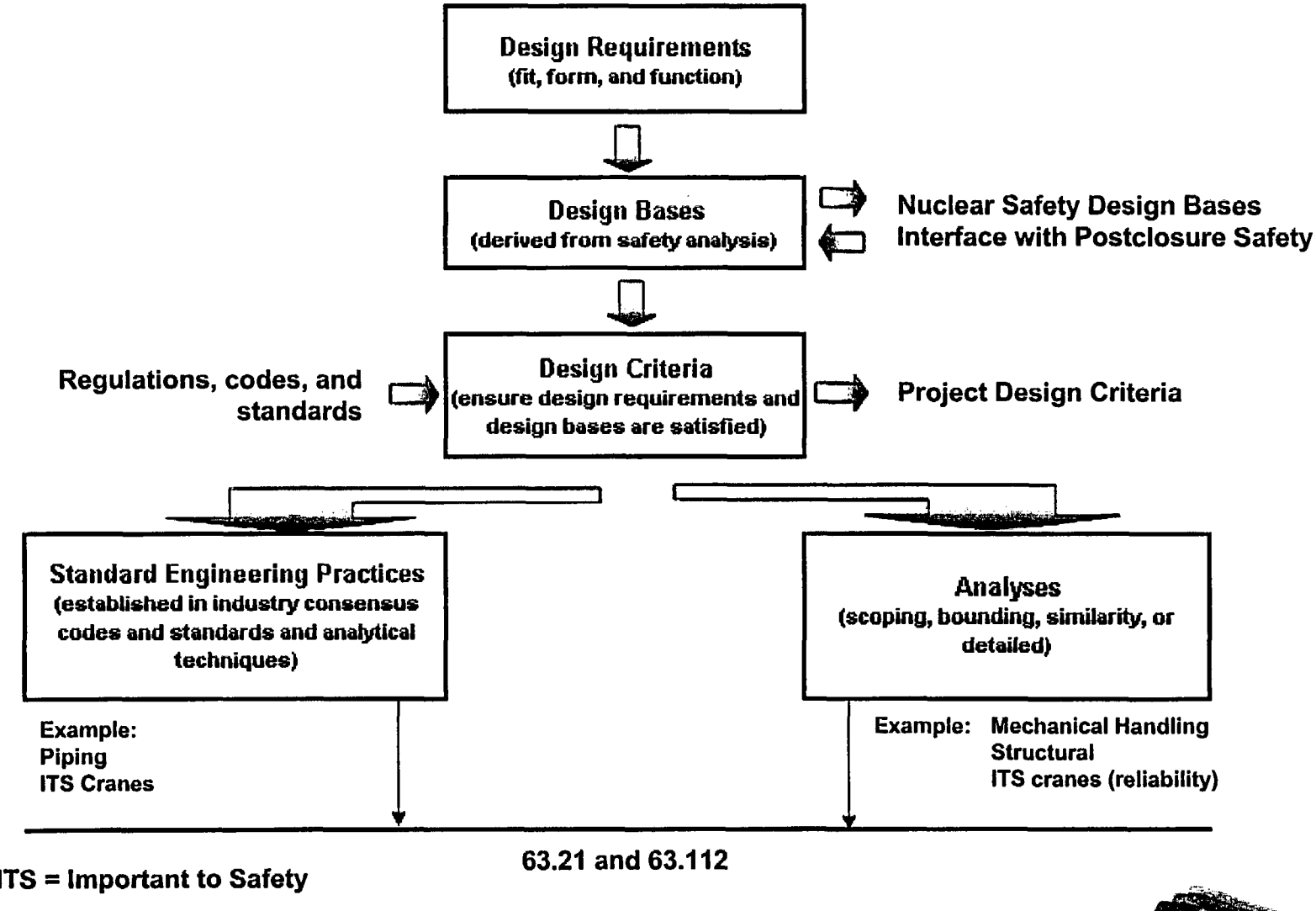
Presented by:
Richard L. Craun
Office of Repository Development
U.S. Department of Energy

July 20, 2005
Las Vegas, Nevada

License Application Overview

- The License Application will contain General Information and a Safety Analysis Report in accordance with 10 CFR 63.21
- The License Application will:
  - Describe surface and subsurface facilities and the engineered barriers
  - Contain sufficient information to demonstrate the ability of structures and systems to perform their intended safety functions in accordance with 10 CFR 63.112

Information Available Process Map



*ITS = Important to Safety

Design Requirements

- Design requirements define the attributes of the structures, systems, and components (SSCs) in terms of fit, form, and function including capabilities, physical dimensions, and any limits
- Design requirements apply to both important to safety (ITS) and non-ITS, SSCs

Design Bases

- The design bases identify the specific safety function to be performed by the structure, system, or component and are contained in the Nuclear Safety Design Bases document
- The design bases identify the values of controlling parameters as reference bounds for design, examples include:
  - Transportation cask and waste package lift heights

Design Criteria

- Design criteria are developed to ensure that design requirements and design bases are satisfied. Design criteria include codes and standards
- Example design criteria sources include:
  - American Society of Mechanical Engineers (ASME) NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)
  - American Concrete Institute (ACI) 318, Building Code Requirements for Structural Concrete
  - Institute of Electrical and Electronic Engineers (IEEE) Standard 141 Recommended Practice for Electrical Power Distribution for Industrial Plants in combination with National Fire Protection Association (NFPA) 70, National Electrical Code

Demonstrated Safety Function

- At the time of License Application submittal, information will be available to demonstrate the structure's and system's ability to perform their intended safety functions in accordance with 10 CFR 63.112(e)(8) using either or a combination of:
  - Standard Engineering Practices
  - Analyses

Standard Engineering Practices

- Industry codes and standards demonstrate the capability to perform credited safety functions to prevent or mitigate the consequences of an event sequence
- Industry consensus codes and standards are identified during the design process and will be specified in the License Application
- Use of a consensus code or standard ensures that the analysis will be performed in accordance with a standardized, acceptable, conservative methodology

Standard Engineering Practices

(Continued)

- **Examples**
  - **ASME B31.3, Process Piping, defines the analytical process for pipe stress analysis, so detailed piping isometrics and pipe stress analysis would not be available for piping systems designed in accordance with this code**
  - **ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), includes specific requirements for cranes. Detailed analyses would not be available for cranes determined to be covered by NOG-1**

Analyses

- For those structures and systems not fully covered by consensus codes and standards, supplemental analyses will be available to demonstrate the capability to perform credited safety functions to prevent or mitigate the consequences of an event sequence
- Analyses may be scoping, bounding, similarity, or detailed
- Any estimates or assumptions that may be used as the basis of the analyses will be conservative
- Use of industry information as the basis for reliability estimates or reliability analysis will be documented

Analyses
(Continued)

- **Examples**
  - **The structural analysis of ITS surface nuclear facilities for seismic ground motions is prepared using multiple lumped mass stick models with soil springs, which have been determined to provide conservative results**
  - **Mechanical handling equipment analyses are performed using consensus codes and standards, as applicable, and include bases for sub-component reliability that are considered in the overall equipment reliability**

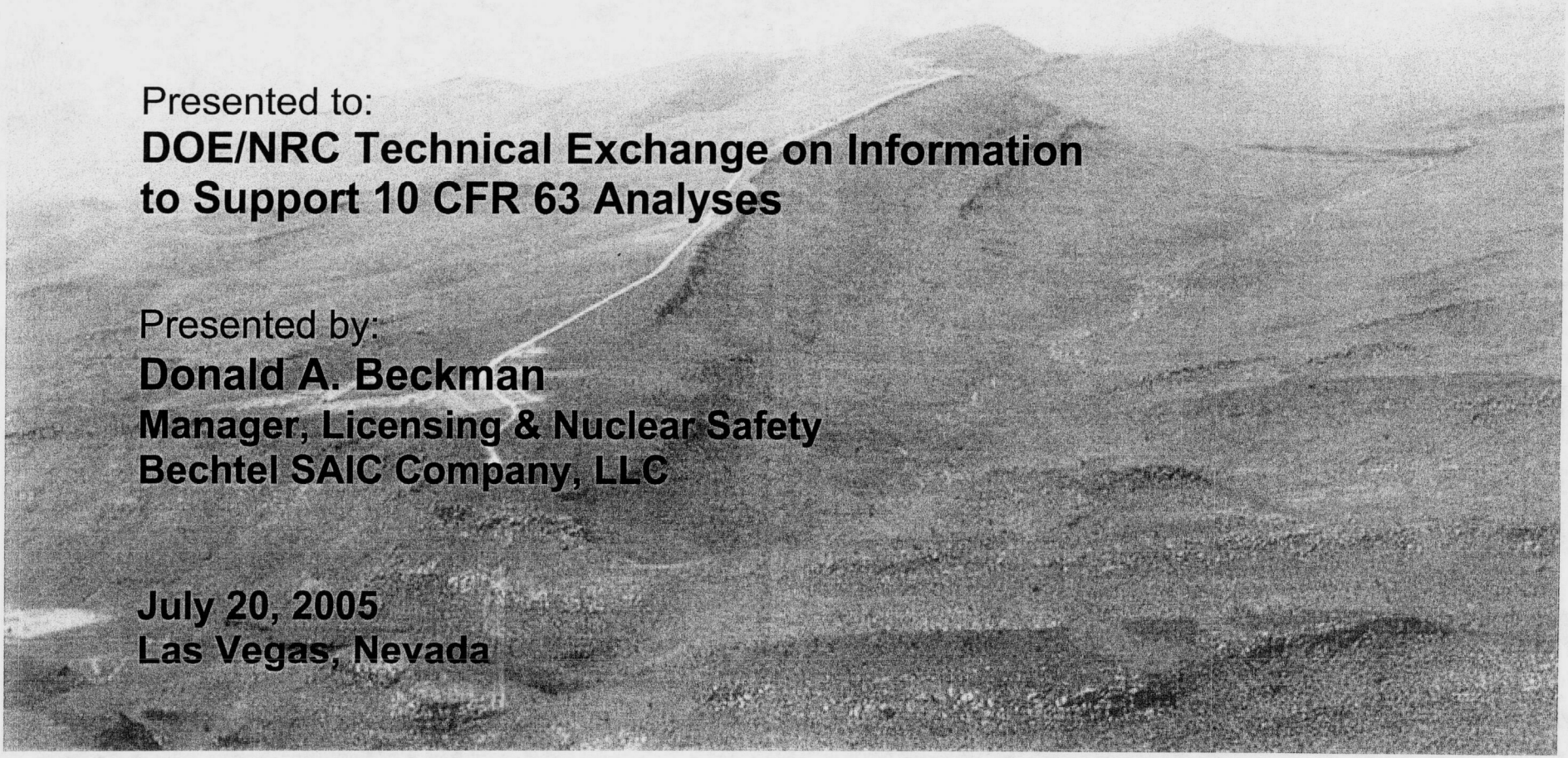
Summary

- The License Application will contain General Information and a Safety Analysis Report in accordance with 10 CFR 63.21(a)
- The License Application will describe surface and subsurface facilities and the engineered barriers in accordance with 10 CFR 63.21(c)(3)
- The License Application or supporting documentation will contain sufficient information to demonstrate the ability of SSCs to perform their intended safety functions in accordance with 10 CFR 63.112(e)(8)

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Licensing Path Forward

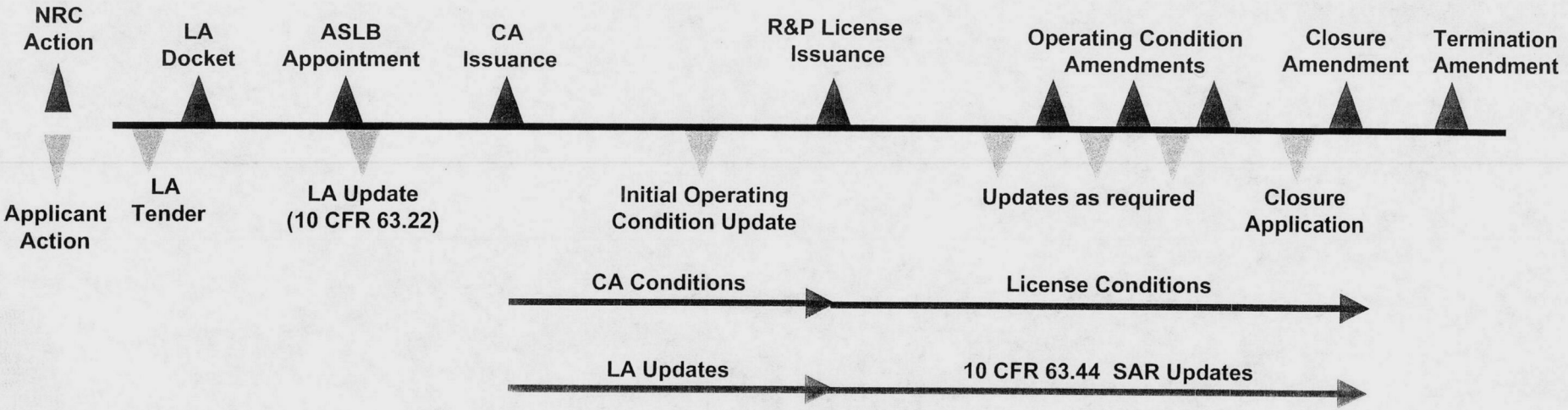


Presented to:
DOE/NRC Technical Exchange on Information to Support 10 CFR 63 Analyses

Presented by:
Donald A. Beckman
Manager, Licensing & Nuclear Safety
Bechtel SAIC Company, LLC

July 20, 2005
Las Vegas, Nevada

Licensing Milestones per 10 CFR 2 and 10 CFR 63



License Application

- Provides description of SSCs, safety analyses, design bases and design criteria that satisfy design bases
- Describes SSCs and operations for 70,000 MTHM inventory

Construction Authorization

- Identifies conditions to satisfy for operation of the repository
- Includes DOE reporting requirements
- Identifies restrictions on subsequent changes to the features of the geologic repository and the procedures authorized

Updates (DOE)

- Provide information to allow initial operations or additional operations
- Provide basis for NRC inspection to lift conditions
- Provide information required by 10CFR 63.46

Amendments (NRC)

- Modifies the R&P License to allow for new conditions or to remove conditions

Summary

- The LA will meet statutory and regulatory requirements
- Supporting detail will be available in technical documents
- Supporting documents will be available for NRC inspection